UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 25, 2017

Mr. Peter A. Gardner
Site Vice President
Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 5.5.11, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NO. MF7359)

Dear Mr. Gardner:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 193 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated February 10, 2016.

The amendment revises Technical Specification 5.5.11, "Primary Containment Leakage Rate Testing Program," to increase the containment integrated leakage rate test program Test A interval from 10 to 15 years.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 193 to DPR-22
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 193
License No. DPR-22

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (NSPM or the licensee), dated February 10, 2016, as supplemented by letters dated October 10 and December 16, 2016, and January 31, February 7, February 16, and March 29, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Renewed Facility Operating License No. DPR-22 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 193, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and prior to the startup from the 2017 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Facility Operating License
and Technical Specifications

Date of Issuance: April 25, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 193

MONTICELLO NUCLEAR GENERATING PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License No. DPR-22 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
5.5-10	5.5-10

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

3. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 2004 megawatts (thermal).

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 193, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

5.5 Programs and Manuals

5.5.10 Safety Function Determination Program (SFDP) (continued)

3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.10.b.1 and 5.5.10.b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.11 Primary Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 2-A, dated October 2008, as modified by the following exceptions:

1. The main steam line pathway leakage contribution is excluded from the sum of the leakage rates from Type B and C tests specified in Section III.B of 10 CFR 50, Appendix J, Option B, Section 6.4.4 of ANSI/ANS 56.8-2002, and Section 10.2 of NEI 94-01, Revision 2-A; and

2. The main steam line pathway leakage contribution is excluded from the overall integrated leakage rate from Type A tests specified in Section III.A of 10 CFR 50, Appendix J, Option B, Section 3.2 of ANSI/ANS 56.8-2002, and Section 8.0 and 9.0 of NEI 94-01, Revision 2-A.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is 56 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 1.2% of containment air weight per day.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated February 10, 2016 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML16047A336), as supplemented by letters dated October 10 and December 16, 2016, and January 31, February 7, February 16, and March 29, 2017 (ADAMS Accession Nos. ML16284A015, ML16355A183, ML17032A038, ML17039A673, ML17048A000, and ML17089A321, respectively), Northern States Power Company (NSPM or the licensee), requested changes to the Technical Specifications (TSs) for the Monticello Nuclear Generating Plant (MNGP).

The supplemental letters dated October 10 and December 16, 2016, and January 31, February 7, February 16, and March 29, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 26, 2016 (81 FR 24663).

The proposed changes would revise the MNGP TS 5.5.11, "Primary Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (ADAMS Accession No. ML003740058), with a reference to Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847), as the implementation document to develop Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," Option B, performance-based primary containment leakage testing program for MNGP. This amendment would allow MNGP to extend its performance-based primary containment integrated leakage rate test (ILRT) or Type A test interval to up to 15 years.

The licensee had previously submitted an amendment request for MNGP to extend the ILRT interval on a one-time basis from 10 years to 15 years in a letter dated April 22, 2002 (ADAMS Accession No. ML021280602). This one-time extension was approved by the U.S. Nuclear Regulatory Commission (NRC or the Commission) as License Amendment No. 134 to Facility Operating License No. DPR-22 for the MNGP, on March 31, 2003 (ADAMS Accession No. ML030440673). The February 10, 2016, license amendment request (LAR) would also delete the listing of this obsolete one-time exception in TS 5.5.11a.1, which was previously granted with Amendment No. 134.

On December 7, 2006, the NRC staff approved full-scope alternative source term license Amendment No. 148 (ADAMS Accession No. ML062790015). In conjunction with Amendment No. 148, the NRC staff granted exemptions to paragraph III.A and Section III.B of 10 CFR Part 50, Appendix J, Option B, which allow the exclusion of main steam line pathway leakage contributions from the overall integrated leakage rate for Type A tests and to the sum of the leakage rates for Type B and Type C tests (ADAMS Accession No. ML062410507). On December 9, 2013, extended power uprate (EPU) License Amendment No. 176 (ADAMS Accession No. ML13316B298) was issued, which increased the maximum licensed thermal power level by approximately 13 percent, from a previous level of 1,775 megawatts thermal (MWt) to 2,004 MWt. As part of Amendment No. 176, the design-basis loss-of-coolant accident (LOCA) containment pressure, P_a, in the plant TSs increased from 42.0 pounds per square inch gauge (psig) to 44.1 psig.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.54(o) require that the primary containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50. Appendix J to 10 CFR Part 50 includes two options: Option A–Prescriptive Requirements, and Option B–Performance-Based Requirements, either of which can be chosen for meeting the requirements of Appendix J.

The testing requirements in Appendix J ensure that leakage through the primary containment and related systems and components penetrating primary containment does not exceed allowable leakage rate value specified in the TSs or associated bases, and integrity of the containment structure is maintained during its service life.

The licensee has adopted and has been implementing Option B for meeting the requirements of Appendix J. Option B of Appendix J specifies the performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate; Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries such as penetrations; and Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and C tests), to ensure integrity of the overall containment system as a barrier to fission product release. The leakage rate test results must not exceed the allowable leakage rate with margin, as specified in the TSs. Option B also requires that a general visual inspection for structural deterioration of the accessible interior and exterior surfaces of the containment, which may affect the containment leaktight integrity, be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires that the RG or other implementation document used by a licensee to develop a performance-based leakage testing program be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in an RG.

The implementation document that is currently referenced in MNGP TS 5.5.11, "Primary Containment Leakage Rate Testing Program," is RG 1.163 (September 1995). RG 1.163 endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995 (ADAMS Accession No. ML11327A025), as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B of 10 CFR Part 50, Appendix J, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

NEI 94-01, Revision 2-A, describes an approach for implementing the optional performance-based requirements of Option B of 10 CFR Part 50, Appendix J. It incorporates the regulatory positions stated in RG 1.163 (September 1995) and includes provisions for extending Type A test intervals to up to 15 years. In the NRC safety evaluation (SE), dated June 25, 2008 (ADAMS Accession No. ML081140105), the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of Option B of 10 CFR Part 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS regarding containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1 of the SE.

Section 50.55a, "Codes and standards," of 10 CFR, contains the containment inservice inspection (CISI) requirements that, in conjunction with the requirements of Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1) of 10 CFR, states, in part, that the licensee:

> ...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience.

The regulations at 10 CFR 50.36(c)(5) require, in part, the inclusion of administrative controls in TSs that are necessary to ensure operation of the facility in a safe manner. This LAR requests a change to a TS under the "Administrative Controls" section of the MNGP TSs.

3.0 TECHNICAL EVALUATION

3.1 Containment Description

The primary containment of MNGP consists of a drywell, a suppression chamber in the shape of a torus, and a connecting vent system between the drywell and suppression chamber. The primary containment is also an enclosure for the reactor vessel, the reactor coolant recirculation system, and other branch connections of the reactor coolant system. The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion, in the shape of a light bulb, and is enclosed in reinforced concrete for shielding purposes. The pressure suppression chamber is a torus-shaped steel pressure vessel located below and encircling the drywell.

The MNGP Updated Final Safety Analysis Report, Revision 33, Section 5, "Containment System," Section 5.2.2.1, "Pressure Suppression System" (ADAMS Accession No. ML16054A419), reads, in part:

> The Primary Containment System, which employs a pressure suppression containment system (constructed of steel), houses the reactor primary vessel, the reactor coolant Recirculation System loops, and other branch connections of the reactor primary system .... The system consists of a drywell, a pressure suppression chamber (wetwell) that stores a large volume of water, a connecting vent system between the drywell and the chamber water pool, isolation valves, ventilating and cooling systems, and other service equipment....
>
> In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure then forces a mixture of non condensible gases, steam, and water through the vents into the pool of stored water in the suppression chamber. The steam condenses rapidly and completely in the suppression pool, resulting in rapid pressure reduction in the drywell.
>
> Non condensible gases forced into the suppression chamber with the steam and water may tend to leave the suppression chamber pressurized with respect to the drywell upon condensation of vapor in the drywell. Vacuum relief valves are provided to prevent such pressurization and the possible accompanying back flow of water from the suppression chamber to the drywell. Cooling systems are provided to remove heat from the drywell, and from the water in the suppression chamber and thus provide continuous cooling of the primary containment under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials which might otherwise be released from the reactor during the course of the accident.

MNGP TS 5.5.11c requires a maximum leakage rate L_a of 1.2 percent of containment air weight per day at the calculated peak pressure, P_a. TS 5.5.11b reads, "The calculated peak containment internal pressure for the design-basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is 56 psig."

3.2 Proposed Changes

MNGP TS 5.5.11a currently states, in part:

> A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, ...

The proposed change will revise TS 5.5.11a to state, in part:

> A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 2-A, dated October 2008, ...

The proposed TS revision deletes 5.5.11a, "exception" 1, which allowed the Type A test frequency to be extended to 15 years for the interval following the test performed in March 1993.

The current TSs contain a typographical error that resulted in separate requirements being numbered 5.5.11a, "exception" 2. The proposed deletion of 5.5.11a "exception" 1 results in proposed TSs with the first current requirement number 5.5.11a "exception" 2 being renumbered 5.5.11a "exception" 1 and the second current requirement number 5.5.11a "exception" 2 remaining 5.5.11a "exception" 2. Additionally in each of these requirements, referenced guidance has been updated to later versions. ANSI/ANS 56.8 is proposed to be changed from 1994 to 2002 and NEI 94-01 from Revision 0 to Revision 2-A.

3.3 Plant-Specific Confirmatory Analysis

The LAR proposes to include reference to NEI 94-01, Revision 2-A, in TS 5.5.11. Therefore, incorporation of the proposed amendment would make the guidance in NEI 94-01 a TS requirement for MNGP. Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 2-A, states that plant-specific confirmatory analyses are recommended when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01, states that the assessment should be performed using the approach and methodology described in Electric Power Research Institute (EPRI) Technical Report (TR)1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (also known as EPRI TR-1018243). The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the SE related to NEI 94-01, dated June 25, 2008, the NRC staff found the methodology in NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML072970208), acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. These conditions, set forth in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submit documentation indicating that the technical adequacy of its probabilistic risk assessment (PRA) is consistent with the requirements of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), relevant to the ILRT extension application.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is

small, consistent with the clarification provided in Section 3.2.4.5[1] of the SE for EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-roentgen equivalent man (rem) per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable, except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the preexisting containment large leak rate accident case (accident case 3b) used by the licensee shall be 100 L_a instead of 35 L_a.

4. An LAR is required in instances where containment overpressure is relied upon for emergency core cooling system (ECCS) performance.

3.3.1 MNGP Specific Risk Analysis

The licensee performed a risk impact assessment for extending the Type A or containment ILRT frequency. For the risk assessment, the risk was evaluated considering the changes from the base case of performing three tests in 10 years to the proposed case of performing one test in 15 years on a permanent basis. The risk assessment was provided in LAR Enclosure 2, "Monticello Nuclear Generating Station: Evaluation of Risk Significance of Permanent Extension of ILRT Extension."

In Section 4.3.1 of Enclosure 1 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in:

1. NEI 94-01, Revision 2;
2. EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals";
3. NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage, Rate Test Surveillance Intervals," October 2001;
4. RG 1.200;
5. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis";
6. Methodology for the Calvert Cliffs Nuclear Power Plant, dated March 27, 2002, to estimate the likelihood and risk implications of corrosion-induced leakage (ADAMS Accession No. ML020920100); and
7. EPRI TR-1009325, Revision 2-A.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, listed in Section 4.2 of the NRC SE. A summary of how each condition has been determined to be met is provided in the sections below.

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[1] Section 4.2 of the Safety Evaluation Report for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is actually provided in Section 3.2.4.6.

Technical Adequacy of the PRA

The first condition stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

Internal Events

Consistent with the information provided in NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," March 22, 2007 (ADAMS Accession No. ML070650428), the NRC staff uses Revision 2 of RG 1.200 to assess technical adequacy of the PRA used to support risk-informed applications. In Section 3.2.4.1 of the Safety Evaluation Report (SER) for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states that Capability Category (CC) I of the American Society of Mechanical Engineers (ASME) PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient for use in the EPRI methodology.

A full scope peer review was performed in April 2013 for the internal events and internal flood PRA model. The peer review was performed using the NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (ADAMS Accession No. ML083430462), as clarified by RG 1.200, Revision 2. The PRA standard provides supporting requirements (SRs) for the PRA against CCs I, II, or III. The peer review resulted in identification of PRA standard supporting requirements that did not meet CC II, or that were met and had related findings. Because CC II is generally an increase in requirements over CC I[2] the NRC staff finds that the peer review against CC II is acceptable for the application. In LAR Enclosure 2, the licensee provided these peer review facts and observations (F&Os) against the PRA standard supporting requirements, its disposition of them, and an assessment of the F&O impact on the ILRT application. The NRC staff reviewed the internal events PRA F&Os and the licensee's dispositions and found that they had no impact on the application, and are, therefore, adequate for the application.

External Events

In Section 3.2.4.2 of the SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events

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[2] The intent of the CC for SRs is that, in general, the degree of scope, level of detail, plant-specificity, and realism increases from CC I to III

> should be performed." This assessment can be taken from existing previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

MNGP employed a plant-specific fire (FPRA) for the ILRT application. A full-scope peer review was performed in March 2015 using the NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," June 2010 (ADAMS Accession No. ML102230070), process and RG 1.200, Revision 2. The peer review F&Os were provided in the LAR, along with the licensee's evaluation for the ILRT application. The licensee stated in its February 7, 2017, letter that a focused-scope peer review was also performed due to FPRA changes.

The licensee evaluated the FPRA F&Os for the ILRT extension in the LAR, Section A.4, and as supplemented in response to NRC staff requests for additional information (RAIs) provided by letter dated February 7, 2017. While some F&Os were on methods, the NRC staff has not performed a review of FPRA methods used in the MNGP FPRA because a Capability Category II FPRA model is not required for the ILRT extension application. The licensee's assessment of the F&Os concluded that there was no impact on the ILRT extension risk analysis.

In response to RAI 1.c, provided by letter dated January 31, 2017, regarding the FPRA model updated after the full scope peer review, the licensee confirmed the FPRA represents the as-built, as-operated plant. There are no plant modifications that are credited in the FPRA that have not been installed in the plant (i.e., no pending credited changes). For the procedure changes that are credited, the procedures were updated prior to the issuance of the FPRA used in the ILRT extension PRA analysis.

The NRC staff finds that the FPRA is adequate for assessing an order of magnitude contribution from fire-related risk for the ILRT extension application because the FPRA has received a full scope and a focused-scope peer review, the licensee has evaluated F&Os for their impact on the ILRT extension application and found no impact, and the FPRA represents the as-built, as-operated plant.

In LAR Enclosure 2, the licensee stated that the individual plant examination of external events (IPEEE) analysis for high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards, resulted in screening these events from further consideration. In response to RAI 1.b, provided by letter dated January 31, 2017, the licensee reviewed changes associated with the IPEEE for these events for their impact on the ILRT extension. The licensee concluded that the risk contribution from these events has no impact on the ILRT extension. For seismic risk, the licensee used the most conservative seismic CDF from the Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (ADAMS Accession No. ML100270582), study, and performed a sensitivity study on the seismic impact for the ILRT extension. The licensee's seismic sensitivity study is discussed in the "Estimated Risk Increase" section below.

The NRC staff concludes that the internal events PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies as it supports Capability Category I of the PRA standard, as clarified by RG 1.200. In addition, the FPRA and other external events risk assessments are sufficient for application to the ILRT extension evaluation as they support an order of magnitude estimate consistent with EPRI guidance. Accordingly, the first condition of the NRC staff's SE for EPRI TR-1009325, Revision 2, is met.

Estimated Risk Increase

The second condition of the NRC staff's SE for EPRI TR-1009325, Revision 2, stipulates that the licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small and consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.5 of the NRC SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percent. Additionally, for plants that rely on containment overpressure (i.e., containment accident pressure[3] (CAP)) for net positive suction (NPSH) for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed further in the "Applicability if Containment Overpressure is Credited for ECCS Performance" section of this SE, MNGP credits CAP. Thus, the associated risk metrics include CDF, LERF, population dose, and CCFP.

By letter dated March 29, 2017, the licensee reported the results of the plant-specific risk assessment in response to RAI 1. The results are for a change in test frequency from three tests in 10 years (the test frequency under 10 CFR Part 50, Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn based on the licensee's risk analyses for the ILRT extension:

1. The increase in total (internal and external) CDF due to loss of CAP for a change in test frequency from three tests in 10 years to one test in 15 years reported by the licensee is 4.52E-7/year (yr). The licensee credited containment cooling for preventing loss of NPSH to the low pressure ECCS pumps for large-break loss-of-coolant accident (LBLOCA) sequences. The NRC staff finds that the licensee did not provide adequate justification for crediting containment cooling to prevent loss of NPSH to the low pressure ECCS pumps. However, the NRC staff estimates that removing this credit would add, at most, 3.58E-7/yr to the change in CDF (see Note 10 to Table 3.3.1-1 below). Therefore, with credit removed for containment cooling for LBLOCA sequences, the change in CDF would be, at most, 8.1E-7/yr. This change in CDF is considered to be "very small" (i.e., below 1E-6/yr) per the acceptance guidelines in RG 1.174.

2. The licensee reported that increase in LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 7.76E-7/yr. This estimate includes both internal events, fire events, and a bounding seismic events sensitivity analysis. The licensee had also considered other potential contributors such as seismic-related loss of CAP, other external events hazards, and steel liner corrosion, and had found them not to be significant contributors. The licensee's estimate also included credit for containment cooling for LBLOCA sequences, and credit for drywell sprays to provide scrubbing and

[3] The NRC staff has discontinued the use of the term "containment overpressure" since the industry uses several definitions of containment overpressure, and the term has been confused with exceeding the design pressure of the containment.

reducing a large early release to a small release, which reduced the estimate for change in LERF due to CAP. The licensee did not provide adequate justification for these two credits; therefore, the NRC staff removed them from the LERF estimate. The NRC estimated the increase in LERF to be 1.29E-6/yr, as shown in Table 3.3.1-2 below. This estimate does not include the seismic bounding sensitivity analysis, but does include the licensee's estimate of seismic contribution from the LAR, which assumed no potential for seismic-induced containment flaw size growth. Sensitivity analyses for seismic hazard impact on containment flaw growth and steel liner corrosion are discussed below and are not expected to cause ΔLERF or the total LERF to be significantly exceeded.

The Class 3b increase in LERF from both CAP and non-CAP risk contributions is close to the RG 1.174 acceptance guideline of 1E-6/yr for "small change" and is not expected to significantly exceed 1E-6/yr. The total LERF from all hazards was reported by the licensee as 9.03E-6/yr, and the NRC staff estimated it as 1.05E-5/yr (Table 3.3.1-1 below). The total LERF is close to the RG 1.174 acceptance guideline of 1E-5/yr and is not expected to significantly exceed 1E-5/yr.

3. Given a change in Type A ILRT frequency from three in 10 years to one in 15 years, and assuming the loss of CAP, the licensee reported the increase in the total population dose was 0.763 person-rem/yr, or 0.898 percent of the total population dose. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2. As discussed in this SE, the licensee included credit for drywell sprays to provide scrubbing and reducing a large early release to a small release, which reduced the estimate for change in LERF due to CAP. The licensee reassigned some core damage sequences to Class 7 and 7a (small releases) instead of Class 3b. The licensee did not provide adequate justification for this reassignment; however, the NRC staff assessed that it does not change the conclusion that the population dose acceptance criteria are met. This is because the dose person-rem values assigned to Classes 3b, 7, and 7a are similar and, therefore, assigning core damage sequences to Class 7 or 7a instead of Class 3b is expected to only have a minimal impact on the estimated person-rem/yr values. Thus, this increase in the total population dose for the proposed change is considered small and supportive of the proposed change.

4. The licensee reported increase in CCFP due to a change in test frequency from three in 10 years to once in 15 years is 0.749 percent. The licensee's reassignment of some Class 3b releases to Class 7 and Class 7a is expected to have a minimal impact on the CCFP calculation because it uses the frequency of those sequences that does not result in containment failure (Class 1 and Class 3a). The CCFP value is below the acceptance guideline of 1.5 percent in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

Table 3.3.1-1: Baseline CDF and LERF/yr

Contributor	CDF	LERF
Internal	8.01E-6[1]	9.20E-7[2]
Seismic	1.9E-5[3]	2.18E-6[4]
Fire	5.01E-5[5]	6.02E-6[5]
Other External Events[7]	< 1E-6[6]	1.15E-7[4]
Class 3b (Non-CAP contribution from Table 2)	N/A	4.83E-7
Class 3b (CAP contribution from internal and fire events)	8.1E-7 (4.52E-7[8] + 3.58E-7[10])	8.1E-7 (4.52E-7[9] + 3.58E-7[10])
Total	7.89E-5	1.05E-5

Notes to Table 3.3.1-1:
1. LAR Enclosure 2, Table 5-1.
2. LAR Enclosure 2, Table 5-2.
3. Generic Issue 199.
4. LAR Enclosure 2, Section 5.3.1: Estimated by multiplying the CDF by the ratio of internal events LERF to internal events CDF.
5. Response to RAI 3.iii (March 29, 2017, letter).
6. LAR Enclosure 2, Section 5.3.1.
7. According to the RAI 1.b response, updated external events considerations remain small and fall within the bounding assessment for external events.
8. Response to RAI 1 (March 29, 2017, letter).
9. ΔLERF = ΔCDF = 4.52E-7 from response to RAI 1 (March 29, 2017, letter).
10. The NRC staff estimates this contribution from removing the internal events PRA LBLOCA containment cooling credit in the response to RAI 1. That is, the LBLOCA contributes 14.5 percent (RAI 4, Reference 6) of internal events contribution to ΔCDF, and 14.5 percent is taken of the ΔCDF of 2.47E-6/yr reported in response to RAI 6.a (February 7, 2017, letter), which includes both internal and fire events. This is a conservative estimate in that it includes more than internal events.

Table 3.3.1-2: Class 3b ΔLERF/yr (3-in-10 to 1-in-15)

Contributor	ΔLERF/yr
Internal	1.74E-8[1]
Fire	2.37E-7[1]
Seismic	1.55E-7[2] Sensitivity: see below
Other External	8.2E-9[3]
SLC	6.51E-8/yr[4] Sensitivity: see below
CAP (internal and fire)	4.52E-7 + 3.58E-7 = 8.1E-7 (See Table 1)
CAP (seismic)	Qualitative assessment[5]
Total	1.29E-6

Notes to Table 3.3.1-2:

1. Response to RAI 3.iv (Reference 7)
2. LAR Enclosure 2, Section 5.3.1: 1.93E-7 (1-in-15 year case) – 3.86E-8 (3-in-10 year case) = 1.55E-7
3. LAR Enclosure 2, Section 5.3.1: 1.02E-8 (1-in-15 year case) – 2.03E-9 (3-in-10 year case) = 8.2E-9
4. RAI 2.a response (Reference 4)
5. See Section 3.2.4 of this SE

Sensitivity Analyses

Seismic Contribution to Class 3b Frequency

In developing the seismic contribution to the Class 3b frequency in the LAR, the licensee scaled the seismic CDF by the ratio of internal event LERF/CDF. Scaling of seismic risk does not take into account consideration that some seismic events may increase the size of a preexisting crack from small to large, thereby adding to the LERF. This increase in the containment flaw size would be a result of the seismic event rather than the accident phenomena, and its associated potential contribution to change in risk would apply to the evaluation of Class 3.

In response to PRA RAI 1.a, provided by letter dated February 7, 2017, the licensee performed a sensitivity analysis. Results showed that if 50 percent of the small leaks were to become large leaks due to the seismic event that the corresponding ΔLERF is estimated to be 2.65E-7/yr, and if all small leaks increased to large leaks, the ΔLERF is estimated to be 4.42E-7/yr. The overall increase in LERF is close to the RG 1.174 guideline for "small" increases of 1E-6/yr as shown in Table 3.3.1-2, however due to the bounding nature of these sensitivity analyses, while this component of LERF risk may contribute to the guideline being slightly exceeded, the NRC staff does not expect the guideline to be significantly exceeded from this contributor.

Containment Steel Liner Corrosion

EPRI TR-1009325, Revision 2-A, recommends a sensitivity analysis to assess the impact of assumptions regarding corrosion-induced leakage of steel containments/liners be completed. The methodology calls for a separate plant-specific assessment of the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended ILRT interval.

The licensee's sensitivity analysis used the Calvert Cliffs Nuclear Power Plant methodology to estimate the risk significance of age-related containment steel liner corrosion large leaks. The method provides an estimate of the likelihood of non-detected containment leakage due to corrosion. This method for assessing ILRT frequency represents a reliability model in which the containment is "as good as found" and not "as good as new" because there is a likelihood that the ILRT may not detect a containment flaw. Thus, undetected flaws could continue to grow until detected and corrected.

To address flaws, limitation and condition number 3 in the NRC staff's SE for NEI 94-01, Revision 2, Section 4.1, requires the licensee to address the areas of the containment structure potentially subject to degradation, including both accessible and inaccessible areas. This limitation and condition references Section 3.1.3 of that SE, which states that plant-specific and generic risk-informed analysis have included specific consideration of degradation in inaccessible areas, and this consideration is based on the availability of that data. In addition, Section 9.2.3.3 of NEI 94-01, Revision 2-A, includes guidance on deficiencies identified during supplemental inspections and consideration of whether or not the extended ILRT interval should continue to be followed. That is, deficiencies identified during supplemental inspections or at any time between ILRTs should be included in the plant's corrective action program, and a determination should be performed to identify the cause of the deficiency and determine appropriate, corrective actions. If the containment performance has degraded (considering leak rates), the unit should be removed from an extended ILRT interval, if applicable, and corrective action pursued. Section 3.4.2 of this SE provides the staff's evaluation of the non-risk portion of this condition.

In response to RAI 2.a, provided by letter dated December 16, 2016, the licensee estimated the Class 3b contribution (i.e., increase in LERF) from steel liner corrosion to be 6.51E-8/yr. The sensitivity analyses increased the likelihood of non-detected containment leakage due to corrosion by orders of magnitude and found that there is significant margin in this likelihood before it could become a significant risk contributor. The overall increase in LERF is close to the RG 1.174 guideline for "small" increases of 1E-6/yr as shown in Table 3.3.1-2; however, due to the bounding nature of these sensitivity analyses, while this component of LERF risk may contribute to the guideline of 1E-6/yr being slightly exceeded the NRC staff does not expect the guideline to be significantly exceeded from this contributor.

Based on the estimated risk results for the risk metrics, the NRC staff concludes that the increase in CDF, and the increase in LERF satisfies the acceptance guidelines of RG 1.174, and the increase in the total population dose and the magnitude of the change in the CCFP for the proposed change are small and supportive of the proposed change. Accordingly, the second condition of the NRC staff's SE for EPRI TR-1009325, Revision 2, is met.

Leak Rate for the Large Preexisting Containment Leak Rate Case

The third condition of the NRC staff's SE for EPRI TR-1009325, Revision 2, states that the methodology in EPRI TR-1009325, Revision 2, is acceptable, except for the calculation of the increase in expected population dose (per year of reactor operation). However, in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the preexisting containment large leak rate accident case (i.e., accident case 3b) used by the licensee shall be 100 L_a instead of 35 L_a.

LAR Enclosure 2, Table 5-3, defines containment failure classes used for the containment Type A ILRT. The containment failure classification of interest for the extended ILRT is Class 3, which is described as independent (or random) isolation failures that include those accidents in which the preexisting isolation failure to seal (i.e., provide a leaktight containment) is not dependent on the sequence in progress. Class 3 sequences include core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components; for example, liner breach or bellows leakage, if applicable. Leaks that are classified as a large early release are designated as Class 3b in the LAR and are evaluated for the risk analysis.

As noted in LAR Enclosure 1, Section 4.3.1, the MNGP analysis used representative containment leakage for Class 3b sequences of 100 L_a based on the guidance provided in EPRI TR-1009325, Revision 2-A, to calculate the increase in population dose for the large leak rate accident case. In subsequent re-analysis of the Class 3b ΔLERF contribution, the licensee re-assigned some Class 3b release sequences to a different release class when the MNGP Level

2 PRA model was employed. In response to RAI 3.i and 3.ii, provided by letter dated March 29, 2017, the licensee explained that a new class (Class 7a) was defined. The licensee stated that Class 7a is a small release reflective of a 100 L_a preexisting leak in containment with release mitigation success following core damage. Class 7 was adjusted accordingly to not include small early releases. Accordingly, Condition 3 of the NRC staff's SE for EPRI TR-1009325, Revision 2, is met.

Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition of the NRC staff's SE for EPRI TR-1009325, Revision 2, stipulates that a LAR is required in instances where containment overpressure is relied upon for ECCS performance. The NRC staff has discontinued the use of the term "containment overpressure" since the industry uses several definitions of containment overpressure, and the term has been confused with exceeding the design pressure of the containment. The inclusion of some or all of the pressure developed in the containment during an accident, in calculating the NPSH, is referred to as CAP. The containment design pressure is never exceeded while crediting CAP.

EPRI TR-1009325, Revision 2, Section 4.2.6, guidance is that the PRA model should be adjusted to account for CAP to evaluate the impacts on CDF and LERF. The combined impacts on CDF and LERF are considered and compared with the risk acceptance guidelines of RG 1.174. The guidance identifies two examples of sequences where CAP may be considered:

- Loss-of-coolant accident scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection in boiling-water reactors (BWRs) or pressurized water reactor sump recirculation, and

- Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long-term use of an injection system from a source inside of containment (for example, BWR suppression pool).

In response to RAI 4.a, provided by letter dated February 7, 2017, which addressed the NRC staff's concern that the licensee's risk assessment provided in LAR Enclosure 2 may not have considered these types of sequences, the licensee revised the risk assessment performed for the LAR by adjusting the internal and FPRA to model loss of CAP. In response to RAI 4.c regarding the NRC staff's concern on the potential for non-minimal cutsets associated with PRA modeling of random and CAP-related failures, the licensee confirmed that inclusion of loss of CAP in the PRA models did not result in non-minimal cutsets.

In response to RAI 1, provided by letter dated March 29, 2017, the licensee further revised the risk assessment. The licensee made further changes to the FPRA model to remove conservatisms related to the time available to establish torus cooling and to establish alternate injection sources that do not depend on the torus, given an initial injection success and using operator actions. The licensee stated that the FPRA model was adjusted to apply the existing operator actions to scenarios where credit could be taken for establishing torus cooling and alternate injection. The licensee also determined that adequate time is available to perform the actions required to establish torus cooling or establish alternate injection with pumps that do not depend on the torus for suction water. Also in response to RAI 1, the licensee made further changes to the internal events PRA model to include containment cooling to retain NPSH for the low pressure ECCS pumps following an LBLOCA. These changes to the internal events and FPRA models contributed to a significant decrease in the CAP-related ΔCDF from 2.47E-6/yr (RAI 6.a response, provided by letter dated February 7, 2017) to 4.52E-7/yr. For the internal events LBLOCA PRA model change, the NRC staff cannot determine that containment cooling credit has been adequately justified without further detailed review. Therefore, the staff does not credit the associated decrease in the CAP CDF with this internal PRA model change in its evaluation of risk provided in the "Estimated Risk Increase" section of this SE.

The NRC staff identified several CAP-related key assumptions and uncertainties as defined in RG 1.200 in RAI 5 (ADAMS Accession No. ML16323A242). In response to RAI 5.b.i, provided by letter dated February 7, 2017, the licensee explained that crediting existing plant conditions impacting NPSH (i.e., the probability of not losing NPSH for the pumps) would preclude the need for CAP as described in the Boiling Water Reactors Owners' Group CAP topical report.[4] In response to RAI 5.c, provided by letter dated January 31, 2017, the licensee also explained that plant Emergency Operating Procedures are used by operators to control pump flow such that NPSH limits are not exceeded and account for CAP or lack thereof. The NRC staff had previously reviewed the referenced TR; however, NRC acceptance of the TR[5] did not include the material in the Section 5.3 of the TR. TR Section 5.3 is related to the CAP-related risk assessment, which credited these two factors. For the ILRT extension risk assessment, the licensee did not include these two factors in the revised risk analysis for loss of CAP.

In response to RAI 5.e, provided by letter dated January 31, 2017, regarding the availability of residual heat removal service water (RHRSW), the licensee explained that upon loss of instrument air, a number of accident mitigation systems for injection or decay heat removal would still be available, including the RHRSW system. In response to RAI 5.f regarding the PRA modeling of loss of offsite power (LOOP) and station blackout (SBO), the licensee explained that loss of CAP risk for LOOP/SBO is decreased due to initiation of suppression pool cooling within a successful time period. Furthermore, if power is not recovered in time to establish suppression pool cooling, the licensee's PRA model accounts for alternate methods of injection and decay heat removal to decrease risk for loss of CAP. In response to RAI 4.b, the licensee identified medium-break LOCA with failure of decay heat removal performed by the residual heat removal system as a dominant internal events CAP risk contributor. Therefore, the NRC staff finds that PRA modeling insights show that the availability of the RHRSW system following a loss of instrument air initiator, residual heat removal system decay heat removal, and the establishment of suppression pool cooling for LOOP/SBO in a successful time window are necessary for decreasing the internal initiating events loss of CAP risk, and that the licensee has adequately addressed the NRC staff's PRA modeling concerns identified in RAI 5.e for the application.

In response to RAI 5.d, the licensee addressed its PRA modeling regarding loss of high pressure injection lube oil cooling, which is also dependent on the suppression pool temperature. The licensee stated that the lube oil cooling failure mode is included in the PRA model because the high pressure injection pumps fail if suppression pool cooling fails, and the licensee determined that no further PRA model changes were necessary for this failure mode. Therefore, the NRC staff finds the licensee has included this failure mode in its PRA modeling for this application.

In response to RAI 5.g, the licensee explained that the containment leak rate assumption used for its risk assessment was consistent with EPRI TR-1009325, Revision 2, Section 4.2.6,

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4 A publicly available version of the topical report is located in ADAMS at Accession No. ML14325A624.

5 Letter from NRC to Boiling Water Reactors Owners' Group, "Final Safety Evaluation for Boiling Water Reactor Owners' Group Topical Report NEDC-33347P/NEDO-33347, Revision 1, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)" (ADAMS Accession No. ML16351A176).

guidance. This guidance permits the assumption that the Class 3b contribution would lead to loss of CAP as a first order of estimate impact approach. Using this approach, the preexisting containment leak rate is assumed to be large enough to result in loss of CAP and to contribute to the change in LERF. In response to RAI 4.b, the licensee confirmed that the leak probability used in the PRA model for the ILRT extension is consistent with the EPRI methodology for calculating a preexisting leak large enough for LERF contribution.

In response to RAI 4, the licensee described the revised approach to evaluate LERF to use the MNGP Level 2 PRA model for internal and fire events. This approach is a change in methodology for calculating the CAP-related ΔLERF because the methodology in the LAR had equated ΔLERF to ΔCDF. In response to RAI 6.a, the licensee reported results that showed a significant decrease in the ΔLERF for loss of CAP. In response to RAI 2, the licensee explained that the Level 2 PRA model credits drywell sprays to provide scrubbing and reduce a large release to a small release. Because of this credit, the licensee reassigned some accident sequences to Class 7 and Class 7a (small release). The NRC staff found that the RAI response did not provide adequate justification for this conclusion. Therefore, for the risk evaluation in this SE, the NRC staff takes the ΔLERF from CAP equal to ΔCDF.

In response to RAI 4.b, the licensee provided a qualitative evaluation on the impact on seismic CAP-related LERF. The licensee stated that if torus cooling is not available with all power failed, the strategy as described in NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," can be used to provide inventory makeup and decay heat removal. In response to RAI 5.a, the licensee stated that MNGP has FLEX pumps that do not depend on the suppression pool as a suction source, and the FLEX pumps are not credited in the current PRA model. In response to RAI 3.iv, the licensee determined that seismic-related loss of CAP is a negligible contributor to the Class 3b ΔLERF. The NRC staff finds that seismic-related loss of CAP was considered in the seismic-reported result in Table 3.3.1-2 and sensitivity analyses for seismic-related flaw growth, both of which considered a pre-existing containment leakage size sufficient to defeat CAP because it was assumed large enough to contribute to LERF, and did not credit FLEX strategy. The seismic significance for the application was discussed previously in this SE without credit for FLEX strategy.

3.3.2 Plant-Specific Confirmatory Analysis Conclusion

Based on its review of the licensee's submittals, the NRC staff concludes that the licensee has demonstrated that the guidance in Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," and Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01, Revision 2-A, has been satisfied. Therefore, the staff finds that the licensee has adequately addressed the plant-specific confirmatory analysis in NEI 94-01.

3.4 Non-Risk Informed Assessment

3.4.1 Containment Inservice Inspection Program

The MNGP overall leaktight integrity of the primary containment is verified through an ILRT (Type A test) as required by 10 CFR Part 50, Appendix J. The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B, and the CISI requirements mandated by 10 CFR 50.55a, together, ensure the continued leaktight and structural integrity of the containment during its service life.

Section 4.4.1 of the LAR stated that the second 10-year CISI interval began on September 9, 2009, and is scheduled to end September 8, 2018. The LAR also noted that the first period of second CISI interval was reduced by 1 year due to the 1-year extension of the first interval.

Section 4.6.1 of the LAR stated that the MNGP CISI program (1) conforms to the applicable requirements of 10 CFR 50.55a and the 2001 Edition with 2003 Addenda of the ASME Boiler and Pressure Vessel Code (Code); and (2) requires examinations of the accessible surfaces of the drywell, torus, vent lines, internal vent system, penetration assemblies and associated integral attachments, pressure retaining bolting, and the drywell interior slab moisture barrier.

In response to the NRC staff's RAI relative to MNGP operating experience and inspection of concrete components, the letter dated October 10, 2016, stated that:

(1) The periodic structural inspections are performed to support implementation of the requirements of the Maintenance Rule, 10 CFR 50.65, and the renewed license aging management program.

(2) The MNGP Structures Monitoring Program requires visual inspection of plant structural features within the scope of the Maintenance Rule on a 5-year interval.

(3) Inspection of the reactor building floor slab, drywell floor slab, shield walls, and reactor vessel pedestal is included in the surveillance.

(4) Based on the inspection reports of the reactor building floor slab, drywell floor slab, shield walls, and reactor vessel pedestal since 1998, the concrete has experienced normal hairline cracking, and certain cracks have been noted to display evidence of groundwater seepage. These cracks have been evaluated for any structural integrity or degradation issues, and no such issues have been identified. The cracks have been repaired, planned to be repaired, or are being monitored for changes.

(5) The site groundwater chemistry is monitored for chemical levels which could be detrimental to the concrete. Because there has been no structural concrete degradation identified in the accessible areas of the plant, no evaluation for possible similar degradation has been performed for inaccessible areas.

(6) In response to NRC Information Notice 2011-20, "Concrete Degradation by Alkali-Silica Reaction" (ADAMS Accession No. 112241029), a petrographic examination of a concrete core sample was performed. The sample chosen was from original construction floor slab in the Turbine Building Condenser Room because it has been periodically infiltrated by ground water. No evidence of alkali-silica reaction or alkali-silica reaction microcracking were observed.

As required by 10 CFR 50.55a(b)(2)(ix)(A), when conditions exist in Class MC accessible areas that could indicate the presence of, or could result in degradation to, Class MC inaccessible areas, the inaccessible areas must be evaluated for acceptability. The LAR stated that no degraded or potentially degraded conditions have been identified or reported for Class MC inaccessible areas as a result of examinations performed as part of the CISI program. In addition, as stated in Section 4.4.1 of the LAR, the licensee (1) has not needed to implement any new technologies to perform inspections of any inaccessible areas at MNGP at this time; (2) actively participates in various nuclear utility owners groups and on ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry; (3) continuously reviews the industry operating experience to determine its applicability to the MNGP; and (4) would explore adjustments to inspection plans and adoption of new, commercially available technologies for examination of inaccessible areas of the containment as they become available.

The LAR stated that to ensure the sand pocket drain line outlets are not obstructed, the plant procedure for filling the reactor well includes a prerequisite to inspect the sand pocket and the air gap drain outlets prior to flooding the refueling area. The LAR also stated that (1) an inspection for leakage from the refueling bellows is performed each refueling outage; (2) MNGP operating history has shown no evidence of refueling seal leakage; and (3) the ongoing IWE inspection and monitoring activities, and the plant features that monitor for leaks past the refueling bellows during refueling (i.e., local light indicator and control room panel alarm), adequately manage aging effects to ensure no loss of intended function.

Section 4.5.3 of the LAR stated that in 1996 and 1997, deterioration in the moisture barrier/caulking between the drywell shell and the concrete basemat was detected. The defective moisture barrier was removed. The drywell shell was found to have minor corrosion and was evaluated as acceptable, and the area was cleaned, recoated, and a new moisture barrier was installed. A portion of the drywell interior basemat concrete in the sand pocket region was excavated to evaluate the exterior of the drywell due to corrosion reported at other licensees due to leakage in the refueling bellows and blocked sand pocket drains. No indication of degradation was detected at the MNGP.

The LAR noted that the most recent IWE examinations were performed during the 2013 Refueling Outage (RFO) 26 in accordance with the requirements of ASME Section XI, Subsection IWE, Table IWE-2500-1, Categories E-A and E-C, for the second period of the second IWE Interval. General visual examination was performed of accessible interior of the drywell. No issues were identified. VT-3 examinations were performed on the disassembled bolted connections for the drywell equipment hatch, the control rod drive hatch, and the drywell head manway. All connections were satisfactory. The moisture barrier at the junction of the drywell base mat to drywell shell was examined, and no evidence of deterioration was identified.

The LAR also stated the following:

a) No new IWE indications were noted during RFO 26 (2013) as a result of coating inspections and general visual inspections in the drywell, vent system (above the waterline) and torus (above the waterline); however, there were some new or changed areas of coating degradation identified in the torus and vent system (downcomers) below the waterline. Rust with minor pitting was reported on seven components (e.g., shell plate) of the torus shell, a legacy arc strike, and a random weld bead were identified during surface preparation for underwater coating repairs. The deepest measured pit noted on the underwater portion of the torus shell was substantially below the pit depth limit. Each of these noted areas were acceptable and were marked for coating repair.

b) Detailed visual (VT-I) augmented examination was performed on downcomers below the water line according to Category E-C, item E4.11 of Table IWE-2500-1, due to degraded coatings and substrate conditions identified during previous outages. The majority of previous indications on the downcomers had no change noted from the previous exam. However, new or changed indications of degraded coatings, larger areas of rust, and new bare substrate were noted on some of the downcomers. Based on visual inspection of the downcomers, none of the indications were severe enough to require pit depth measurement. Corrosion in the vent system, as in previous outages, was negligible. The downcomers will continue to be examined, as required under Category E-C, item E4.11.

c) Based on previous and the most recent coating inspection performed during RFO 27 (2015), there is no evidence of systemic coating failure from aging or other systemic causes. Areas of degraded coatings with related corrosion-like indications, such as rust staining and surface rust, have been reported in the high humidity or wetted areas on the internal surfaces of the torus and the portion of the vent system downcomers near or below the waterline. The coating in the underwater portion of the torus is in substantially the same as-left condition as in RFO 23 (2007). Indications are evaluated each outage, including minor pitting, and have been determined to be acceptable. No conditions indicative of potential degradation in inaccessible areas have been identified. Numerous areas with degraded coatings have been recoated, including extensive coating repairs to the interior of the torus shell while drained in the 2007 RFO, and limited underwater coating repairs were made during the 2013 RFO. The next underwater inspection of coatings is planned for RFO 28 (2017) and coincides with the third period of the second interval of the IWE examinations.

Based on the results of the MNGP recent IWE inspections and inspection of concrete components discussed above, the NRC staff finds that there has not been evidence of significant degradation of the MNGP primary containment, and the degradations noted have been entered into the MNGP corrective action program and appropriately managed and/or corrected.

Based on the NRC staff's evaluation of the licensee's submittal of February 10, 2016, and supplemental letter dated October 10, 2016, the NRC staff concluded that the age-related degradation and structural integrity of the primary containment are adequately managed, and the results of the CISI demonstrate acceptable performance of the MNGP primary containment. The MNGP primary containment will continue to be periodically inspected, as required by 10 CFR 50.55a and the ASME Code, according to the CISI program, if the current Type A test interval is extended from 10 years to 15 years. Therefore, the NRC staff concluded that the MNGP CISI results and operating experience support the extension of the Type A test interval.

3.4.2 MNGP Type A Test Performance History

Per TS 5.5.11, MNGP, the maximum allowable containment leakage rate L_a is 1.2 percent of containment air weight per day at the calculated peak pressure, P_a. From TS 5.5.11b, the calculated peak containment internal pressure for a design-basis LOCA, P_a, of 44.1 psig. The containment design pressure is 56 psig.

Since the completion of the Type A test in May 1980, a total of four more ILRTs have been performed on the MNGP containment, all with "as found" satisfactory results. These four ILRT test results were documented in LAR Enclosure 1, Section 4.2.4. These test results are summarized in Table 3.4.2-1 below, as amended with the licensee's response to RAI 2, provided by letter dated October 10, 2016:

Tabe 3.4.2-1: MNGP Type A ILRT Results at P_a[1]

Date	As Found Leak Rate (% weight/day)	As Found TS Acceptance Criteria	As Left Leak Rate (% weight/day)	As Left TS Acceptance Criteria
December 1984	0.7222	1.2	0.5484	0.9
October 1989	0.6183	1.2	0.5354	0.9
March 1993	0.8240	1.2	0.3943	0.9
April 2007[2]	0.7501[3]	1.2	0.6503[3]	0.9

(1) All performances of the ILRT were performed at full test pressure, P_a, established as the licensing basis as of the date of performance.
(2) P_a increased from 42.0 psig to 44.1 psig with EPU (License Amendment No. 176).
(3) Revised value per RAI 2 to correct licensee transcription errors.

The NRC staff notes that the last sentence of Section 9.2.3, "Extended Test Intervals," of NEI 94-01, Revision 2-A, states: "In the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR Part 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure (P_a)." Section 9.1.2 of the NE 94-01, Revision 2-A, reads in part: "The elapsed time between the first and the last tests in a series of consecutive passing tests used to determine performance shall be at least 24 months."

As displayed by the as-found ILRT test data and footnotes of Table 3.2.4-1, the P_a requirement of NEI 94-01, Revision 2-A, Section 9.2.3, is satisfied since these four ILRTs were performed at full test pressure, P_a. In addition, the elapsed time limitation of Section 9.2.1 is satisfied by the time lapse between the two ILRTs of March 1993 and of April 2007.

The MNGP Appendix J, Option B, current license basis TS 5.5.11a references RG 1.163. The "Regulatory Position" of RG 1.163 invokes the guidance of NEI 94-01, Revision 0. Section 9.2.3 of NEI 94-01, Revision 0, reads, in part:

> For purposes of determining an extended test interval, the performance leakage rate is determined by summing the UCL [upper confidence limit] (determined by containment leakage rate testing methodology described in ANSI/ANS 56.8-1994) with as-left MNPLR [minimum pathway leakage rate] leakage rates for penetrations in service, isolated or not lined up in their accident position (i.e., drained and vented to containment atmosphere) prior to a Type A test. In addition, any leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. If the leakage can be determined by a local leakage rate test, the As-left MNPLR for that leakage path must also be added to the Type A UCL. ...
>
> In reviewing past performance history, Type A test results may have been calculated and reported using computational techniques other than the Mass Point method from ANSI/ANS-56.8–1994 (e.g., Total Time or Point–to–Point). Reported test results from these previously acceptable Type A tests can be used to establish the performance history. Additionally, a licensee may recalculate past Type A UCL (using the same test intervals as reported) in accordance with

> ANSI/ANS-56.8–1994 Mass Point methodology and its adjoining Termination criteria in order to determine acceptable performance history.

NEI 94-01, Revision 2-A, Section 9.2.3, reads similarly, except the test standard invoked is ANSI/ANS-56.8–2002. To this end, the NRC staff requested additional information about the "As-found" Type A test results of April 2007. In particular, the NRC staff requested that the licensee provide a detailed and comprehensive breakdown of the test-specific data for the MNGP containment "as-found" leakage rate of 0.7323 percent primary containment air weight per day. The licensee responded by letter dated October 10, 2016, and provided a comprehensive breakdown of the ILRT results of April 2007. The licensee did note that transcription errors occurred during the transfer of data between LLRT procedures and ILRT procedure. The licensee provided revised "as-found" and "as-left" ILRT leakage rate values to correct the erroneous LAR values. These revised values are reflected in Table 3.4.2-1 above. In addition, the RAI response provided detailed information about the minimum containment test pressure recorded during the 2007 ILRT. The minimum containment pressure in psig for the 2007 ILRT was determined to be 42.445 psig, which exceeded the then relevant pre-EPU P_a value of 42 psig.

The NRC staff notes that ANSI/ANS-56.8-1994, paragraph 3.2.11, "Type A Test Pressure," reads in part:

> The Type A test pressure shall not be less than 0.96 P_{ac}. nor exceed P_d ... The test pressure shall be established relative to the external pressure of the primary containment measured at the start of the Type A test.

P_{ac} equals peak accident pressure. Therefore, P_{ac} is synonymous with P_a. ANSI/ANS-56.8-2002, paragraph 3.2.12, "Type A Test Pressure," conveys the same information. Therefore, the lower 2007 ILRT test bound of 0.96Pac for MNGP primary containments equaled 40.32 psig.

P_d equals containment design pressure. The licensee did not provide a value for the maximum test pressure recorded during the 2007 ILRT. For the MNGP primary containment, P_d equals 56 psig. During 2007 the differential pressure between P_{ac} and P_d for the MNGP primary containment equaled 14 psid. Therefore, due to the constraints of time, procedural controls, and the additional resources needed to attain a test pressure in excess of P_d, it is reasonable to conclude that P_d was not exceeded during the 2007 ILRT. Accordingly, the NRC staff concluded that the 2007 ILRT minimum recorded test pressure of 42.445 psig satisfied the test methodology requirements of ANSI/ANS-56.8-1994, paragraph 3.2.11.

While not a regulatory requirement, the licensee concluded, and the staff concurs, that the minimum recorded test pressure of 42.445 psig also satisfies the post EPU P_a value of 44.1 psig, as allowed by ANSI/ANS-56.8-2002.

The NRC staff notes that Section 9.2.3 states that a licensee may recalculate past Type A test results to demonstrate conformance with the definition of "performance leakage rate" contained in NEI 94-01, Revision 2-A. The NRC staff also notes that the ILRT results from 1984, 1989, and 1993, demonstrated ample margin (i.e., ≥ 31 percent) between each ILRT value and L_a. Accordingly, the NRC staff did not request in RAI 2 that the licensee reconstitute the Type A test results conducted during RFOs 20-30 years in the past. Based on the demonstrated ample margin, the NRC staff found the licensee's response to RAI 2 acceptable.

TS 5.5.11d.1 establishes the maximum limit for MNGP startup following completion of Type A testing at $\leq$ 0.75 L_a, which equals 0.90 percent of containment air weight per day.

The MNGP containment was designed for a leakage rate L_a not to exceed 1.2 percent by weight of containment air per day at the calculated peak pressure, P_a. As displayed in Table 3.2.4-1, there has been adequate margin to the "as found" performance limit as described in TS 5.5.11c of L_a equal to 1.2 percent weight/day for the four most recent historical ILRTs.

Past MNGP ILRT results have confirmed that the containment leakage rates are acceptable with respect to the design criterion of 1.2 percent leakage of containment air weight (L_a) per day at the design-basis LOCA pressure (P_a). Since the last two Type A tests for MNGP had "as-found" test results of less than 1.0 L_a, a test frequency of 15 years, in accordance with NEI 94-01, Revision 2-A, would be acceptable for MNGP.

Based on the historical MNGP ILRT test results and the licensee's response to RAI 2, the NRC staff concluded that the guidance in Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 2-A, has been satisfied.

3.4.3 MNGP Type B Test and Type C Test Performance History

The NRC staff reviewed the local leak rate summaries from the last five RFOs contained in the table entitled, "MNGP Type B and C LLRT Combined As-Found / As-Left Trend Summary," of LAR Enclosure 1, Section 4.4.3, "Primary Containment Leakage Rate Testing Program - Type B and Type C Testing Program."

The licensee provided a response to NRC RAI 3 that provided additional detail about the LLRT results contained in the table in the letter dated October 10, 2016. The licensee indicated that RFO 26, during spring 2013, was the last RFO completed before the EPU license amendment (Amendment No. 176) was issued in December 2013. P_a increased from 42 psig to 44.1 psig as a result of EPU implementation. This increase in P_a resulted in an increase in L_a from 458.6 standard cubic feet per hour (scfh) to 475.1 scfh. During RFO 27 in 2015, all LLRT tests performed were with the higher P_a. By the end of RFO 27, approximately 80 percent of the Type B and Type C components had been tested at the higher post-EPU P_a. It was noted that per MNGP's Operating License Condition 14, leak rate tests performed under the pre-EPU test conditions, are not required to be performed at EPU conditions until their next scheduled performance. The licensee noted that the remaining Type B and Type C components are scheduled for local leak rate testing at the elevated P_a during RFO 28 in 2017.

The data contained in the table supported the following conclusions:

- The "as-found" minimum pathway leak rate average for MNGP shows an average of 31.3 percent of 0.6 L_a with a high of 42.9 percent of 0.6 L_a (i.e., 0.258 L_a).

- The "as-left" maximum pathway leak rate average for MNGP shows an average of 57.2 percent of 0.6 L_a with a high of 75.1 percent of 0.6 L_a (i.e., 0.451 L_a).

Based on the review of the data contained in the table entitled, "MNGP Type B and C LLRT Combined As-Found / As-Left Trend Summary," the NRC staff concluded that the aggregate results of the "as-found minimum pathway" and "as-left maximum pathway" for all the Type B and Type C tests from 2007 through 2015 demonstrate a history of successful tests, since the aggregate test results were significantly less than the Type B and Type C test TS limit of

< 0.60 L_a contained in TS 5.5.11d.1. Furthermore, the table shows that there has been no as-found failure that resulted in exceeding the TS 5.5.16.d.1 limit of 0.6 L_a (285 scfh) and demonstrates a history of successful tests. The as-found minimum pathway summations represent the high quality of maintenance of Type B and Type C tested components, while the as-left maximum pathway summations represent the effective management of the containment leakage rate testing program by the MNGP program owner.

In addition, LAR Section 4.4.3 provided statistics on the number of Type B components and Type C containment isolation valves (CIVs) on the maximum allowed extended frequencies of 120 months and 60 months, respectively. The total number of Type B and Type C components eligible for extended intervals equals 128. Of the eligible Type B penetrations and Type C CIVs, more than 91 percent of the cumulative population was on extended test intervals. LAR Section 4.4.4, "Type B and Type C Local Leak Rate Testing Program Implementation Review," indicates that there were no components on extended LLRT intervals that exhibited unacceptable performance (i.e., no administrative limit failures) during the previous two RFOs of 2013 (RFO 26) and 2015 (RFO 27).

The NRC staff concluded that from the percentage of Type B and Type C components on extended frequencies and from the lack of administrative failures during RFO 26 and RFO 28, the MNGP primary containment leakage rate testing program represents good performance. This further supports the proposed changes to TS 5.5.11a.

In summary, the NRC staff concluded that the cumulative Type B and Type C test results were below the acceptance limit of TS 5.5.11d.1.

Therefore, the NRC staff finds that the licensee is effectively implementing the Type B and Type C leakage rate test program, as required by Option B of 10 CFR Part 50, Appendix J.

3.4.4 NRC Safety Evaluation for NEI 94-01, Limitations and Conditions Assessment

As required by 10 CFR 50.54(o), the MNGP containment is subject to the requirements set forth in 10 CFR Part 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the MNGP 10 CFR Part 50, Appendix J, testing program plan is based on RG 1.163, which endorses NEI 94-01, Revision 0. The LAR, as supplemented, proposes to revise the MNGP 10 CFR Part 50, Appendix J, testing program plan by implementing the guidance contained in NEI 94-01, Revision 2-A.

By letter dated June 25, 2008, the NRC published an SE with limitations and conditions for NEI 94-01, Revision 2 (ADAMS Accession No. ML081140105). In the SE, the NRC staff concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR Part 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TSs regarding containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of the SE. Section 4.1 of the SE establishes limitations and conditions pertaining to deterministic requirements; while Section 4.2 establishes limitations and conditions pertaining to the plant's PRA analysis. More explicitly, the SE included provisions for extending the ILRT Type A interval to a maximum of 15 years on a permanent basis, subject to the six limitations and conditions provided in the SE. The NRC staff noted in the SE that NEI 94-01, Revision 2, incorporates the regulatory positions stated in RG 1.163. The accepted version of NEI 94-01, Revision 2, was subsequently issued as Revision 2-A to incorporate the June 25, 2008, NRC SE.

Type B testing ensures that the leakage rate of individual containment penetration components is acceptable. Type C testing ensures that individual containment isolation valves are essentially leaktight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

The LAR proposes that MNGP invoke NEI 94-01, Revision 2-A, and ANSI/ANS 56.8–2002 as the reference documents for the MNGP primary containment leakage rate testing program in TS 5.5.11. Accordingly, the LAR does not request to extend the frequencies of the Type B or Type C test intervals.

The NRC staff has found that the use of NEI 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided the following applicable six limitations and conditions from Section 4.1 of the SE are satisfied:

Limitation/Condition 1

Limitation/Condition 1 stipulates that for calculating the Type A leakage rate, the licensee should use the definition in the NEI 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002. Limitation/Condition 1 is discussed on page 51 of Enclosure 1 to the LAR.

The table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions," in Section 4.7.1 of the LAR, states, in part, that, "NSPM will utilize the definition in NEI 94-01, Revision 2-A, Section 5.0," and included a reference to footnote 15. Footnote 15 states, "As part of the amendment implementation process the definition in NEI 94-01, Revision 2-A, will be adopted for calculating the Type A leakage rate."

The NRC staff reviewed the definition of "performance leakage rate" contained in NEI 94-01, Revision 2, and Revision 2-A. The NRC staff concluded that the definition contained in both is identical.

Therefore, the NRC staff finds that NSPM will use the definition found in Section 5.0 of NEI 94-01, Revision 2, for calculating the Type A leakage rate in the MNGP primary containment leakage rate testing program.

Based on the above review, the NRC staff finds that the licensee has adequately addressed Limitation/Condition 1.

Limitation/Condition 2

Limitation/Condition 2 stipulates that the licensee should submit a schedule of containment inspections to be performed prior to and between Type A tests.

The table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions," in Section 4.7.1 of the LRA states, "A projected schedule for containment inspections is provided in Subsection 4.4.5 of this enclosure."

The NRC staff's SE, Section 3.1.1.3, for NEI 94-01, Revision 2, states in part that:

> NEI TR 94-01, Revision 2, Section 9.2.3.2, states that: "To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years." NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3 of RG 1.163, are acceptable considering the longer 15 year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.

NEI 94-01, Revision 2-A, Section 9.2.3.2, "Supplemental Inspection Requirements," states, in part, that:

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

The NRC staff reviewed LAR Enclosure 1, Subsection 4.4.1, "Containment Inservice Inspection Plan (IWE Plan)." The IWE program was developed with a first interval start date of September 9, 1998.

The three inspection periods during the second inspection interval are as follows:

First Period:	September 9, 2009 – September 8, 2011
Second Period:	September 9, 2011 – September 8, 2015
Third Period:	September 9, 2015 – September 8, 2018

The first inspection period during the third inspection interval is as follows:

First Period:	September 9, 2018 – September 8, 2021

Currently, TS 5.5.11a requires, in part, visual examinations in accordance with the guidelines contained in RG 1.163. Regulatory Position 3 of this RG requires that these examinations should be conducted prior to initiating a Type A test.

The LAR states in the MNGP response that the containment examination requirements will be maintained in accordance with the projected schedule as provided in LAR Enclosure, Subsection 4.4.5, "Supplemental Inspection Requirements." The NRC staff's review of the table entitled "ILRT Versus General Visual (IWE and Appendix J) and Coatings Examination" indicates that four "General Visual Examinations" of the MNGP "Containment Interior and Exterior Surfaces – Drywell and Torus" since the completion of the last ILRT during April 2007 (RFO 23) and the start of a 15-year extended interval ILRT of 2021 (RFO 30).

Based on the schedules contained in LAR Enclosure, Subsections 4.4.1 and 4.4.5, MNGP will meet the requirements of the proposed revision to TS 5.5.11; the inspection requirements of ASME Code Section XI, Subsection IWE; and NEI 94-01, Revision 2-A, Section 9.2.3.2.

Based on the above review, the NRC staff finds that the licensee has adequately addressed Limitation/Condition 2.

Limitation/Condition 3

Limitation/Condition 3 stipulates that the licensee address the areas of the containment structure potentially subjected to degradation.

The LAR table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions" in Section 4.7.1, states "Refer to Subsections 4.4.1 and 4.4.5 of this enclosure" for this item.

The NRC staff's SE Section 3.1.3, for NEI 94-01, Revision 2, reads, in part:

> In approving for Type A tests the one-time extension from 10 years to 15 years, the NRC staff has identified areas that need to be specifically addressed during the IWE and IWL inspections including a number of containment pressure-retaining boundary components (e.g., seals and gaskets of mechanical and electrical penetrations, bolting, penetration bellows) and a number of the accessible and inaccessible areas of the containment structures (e.g., moisture barriers, steel shells, and liners backed by concrete, inaccessible areas of ice condenser containments that are potentially subject to corrosion).

The NRC staff reviewed the following LAR sections and subsections in evaluating the licensee's response to this Limitation/Condition:

- 4.4.1, "Containment Inservice Inspection Plan (IWE Plan)"
- 4.4.2, "Nuclear Coatings Program"
- 4.4.5, "Supplemental Inspection Requirements"
- 4.5, "Operating Experience"
- 4.6, "License Renewal Aging Management"

The NRC staff notes that LAR Enclosure 1, page 28 of 64, "Containment Examination," reads, in part:

> General Visual examination includes 100 percent of the accessible surface areas during each Inservice Inspection Period and requirements as modified by 10 CFR 50.55a.

- General Visual (VT-3) examination of 100 percent of wetted surfaces of submerged areas is performed each interval.
- General Visual (VT-3) examination of 100 percent of BWR vent system accessible surface areas is performed each interval.
- General Visual examination to include moisture barrier materials intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining metal containment shell or liner at concrete-to-metal interfaces and at metal-to-metal interfaces which are not seal welded. Containment moisture barrier materials include caulking, flashing and other sealants used for this application. Also, 100 percent examination is required during each Inspection Period. Examination is performed to identify tears, cracks, or damage that permits moisture intrusion.
- General Visual examination performed each period for E1.11 includes pressure retaining bolted connections. Once each interval, the pressure retaining bolted connections require examination using VT-3. Upon disassembly VT-3 examination is required.

LAR Enclosure 1, page 27 of 64, "Inaccessible Areas," reads, in part:

> As required by 10 CFR 50.55a(b)(2)(ix)(A)(1), when conditions exist in Class MC accessible areas that could indicate the presence of, or result in degradation to Class MC inaccessible areas, the inaccessible areas must be evaluated for acceptability. No degraded or potentially degraded conditions have been identified or reported for inaccessible Class MC areas as a result of examinations for the CISI Plan.

Furthermore, LAR Subsection 4.4.1 indicates that ASME Code Class MC boundaries subject to examination are the containment structure and connecting penetrations, appurtenances, and parts that form the primary containment leaktight boundary. These include the drywell shell, the drywell head assembly, penetration connections, penetration sleeves of electrical penetrations, various equipment and personnel hatches, and Class MC component supports.

Based on this review, the NRC staff finds that the licensee has addressed, the issues of SE Section 3.1.3 and Limitation/Condition 3.

Limitation/Condition 4

Limitation/Condition 4 stipulates that the licensee address any tests and inspections performed following major modifications to the containment structure, as applicable.

The table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions," in Section 4.7.1 of the LAR states, "There are no major modifications planned for the MNGP that would affect the containment structure."

The NRC staff's SE, Section 3.1.4, for NEI 94-01, Revision 2, indicates:

> Section 9.2.4 of NEI TR 94-01, Revision 2, states that: "Repairs and modifications that affect the containment leakage integrity require LLRT [local leak rate test] or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation."

> Article IWE-5000 of the ASME Code, Section XI, Subsection IWE (up to the 2001 Edition and the 2003 Addenda), would require a Type A test after major repair or modifications to the containment. In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure.

The LAR states in the MNGP response that there are no major containment modifications planned. Furthermore, LAR Section 4.4, "Non-Risk Based Assessment," page 31 of 64, reads:

> If repair/replacement activities of ASME Section XI, Subsection IWA-4000 become necessary on Class MC components, as authorized by the Fifth Interval ISI Plan Relief Request RR-007 ... , post repair/replacement pressure test requirements for components and parts of the pressure retaining boundary shall comply with the requirements of the 2007 Edition including the 2008 Addenda of ASME Section XI, Subsection IWE-5000, as well as all applicable conditions in 10 CFR 50.55a for post-repair/replacement pressure testing of Class MC components.

The NRC staff noted in RAI 1 that the above LAR excerpts are "forward looking" with respect to plans for any future containment modification. The NRC staff requested that the licensee provide evidence of a historical consistency with guidance of SE Section 3.1.4. By letter dated October 10, 2016, the licensee provided a response to RAI 1. The response listed seven significant modifications made to the MNGP containment prior to performance of the ILRT during 2007. The entire listing reads:

- Modifications were performed to the pressure suppression system to address generic Mark I Containment Load Program conclusions.
- Modifications were performed to permit proper Type C testing of valves where the testing did not conform to Appendix J.
- A new penetration was added for installation of the Hard Pipe Vent System in response to Generic Letter 89-16, "Installation of a Hardened Wetwell Vent."
- Upgrades to the Drywell Personnel Airlock equalizing valves and electrical penetration.
- The Reactor Pressure Vessel Head Spray lines were cut and capped as this function was removed.
- The Combustible Gas Control System lines communicating with the primary containment were cut and capped as part of the removal of the system from service with the adoption of 10 CFR 50.44 rule changes allowing removal of the hydrogen recombiners.
- The primary containment double ply bellows assembly for penetration X-16B, "A Core Spray," was replaced with a modified bellows design in 1998.

The response indicated all seven of these containment modifications were tested in accordance with 10 CFR Part 50, Appendix J, and ASME XI, Subsection IWE, as applicable, in accordance with the requirements of the MNGP design modification process.

Based on MNGP's past performance, as reflected in the response to RAI 1, the NRC staff finds that the licensee has adequately addressed the issues of SE Section 3.1.4 and Limitation/Condition 4.

Limitation/Condition 5

Limitation/Condition 5 of stipulates that the normal Type A test interval should be less than 15 years. If a licensee has to use the provisions of Section 9.1 of NEI 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.

The table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions," in Section 4.7.1 of the LAR, states:

> NSPM will follow the requirements of NEI 94-01, Revision 2-A, Section 9.1. In accordance with the guidance of Section 3.1.1.2 of the SE for NEI 94-01, Revision 2-A, as further amplified by the additional guidance provided in RIS 2008-27.[16] NSPM will demonstrate to the NRC staff that an emergent unforeseen condition arose if an extension beyond the 15-year ILRT interval became necessary.[17]

Footnote 16 states, "RIS 2008-27, 'Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50'." Footnote 17 states, "As part of the amendment implementation process the program document controlling Type A testing will be revised to reflect this limitation."

Section 3.1.1.2 of the NRC staff's SE, dated June 25, 2008, reads:

> Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

The NRC staff notes that the licensee has acknowledged the guidance of NEI 94-01, Revision 2-A, SER Section 3.1.1.2, and accepted the NRC staff's position discussed in Limitation/Condition 5. By referencing Regulatory Issue Summary 2008-27, the licensee has confirmed its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and that any requested permission (i.e., for such an extension) will demonstrate to the NRC staff that an unforeseen emergent condition exists.

Based on the above review, the NRC staff finds that the licensee has adequately addressed Limitation/Condition 5.

Limitation/Condition 6

Limitation/Condition 6 stipulates that for plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, including the use of past containment ILRT data.

The table entitled "NEI 94-01, Revision 2-A, Limitations and Conditions" in Section 4.7.1 of the LAR states, "Not applicable. MNGP was not licensed under 10 CFR Part 52."

The NRC staff finds that MNGP operates according to the Renewed Facility Operating License No. DPR-22 and is not licensed under the requirements of 10 CFR Part 52. Therefore, Limitation/Condition 6 does not apply.

Limitation/Condition Summary

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed the six conditions identified in Section 4.1 of the NRC SE for NEI 94-01, Revision 2-A. Therefore, the NRC staff finds it acceptable for MNGP to adopt NEI 94-01, Revision 2-A, as the implementation document in TS 5.5.11, "Primary Containment Leakage Rate Testing Program."

3.4.5 Non-Risk-Based Evaluation Conclusion

The NRC staff reviewed the Type A, Type B, and Type C leakage test results related to the licensee's proposal to extend 10 CFR Part 50, Appendix J, Type A, test interval from 10 years to 15 years on a permanent basis.

The ILRT results provided in LAR Section 4.2.4, "MNGP Integrated Leakage Rate Testing History," indicate that the last two consecutive Type A tests at MNGP were successful with the "as-found" containment performance leakage rates less than the maximum allowable containment leakage rate of 1.2 percent containment air weight per day (i.e., 1.0 L_a at P_a) contained in TS 5.5.11d.1. Therefore, the NRC staff finds that the performance history of Type A tests supports extending the current ILRT interval on a permanent basis to 15 years as permitted by NEI 94-01, Revision 2-A.

Based on the NRC staff review of the licensee's submittal of February 10, 2016, supplemental information provided in the RAI response letter of October 10, 2016, and the regulatory and technical evaluations above, the NRC staff finds that there is reasonable assurance that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 2-A as the 10 CFR Part 50, Appendix J, Option B, implementation document.

The NRC staff also finds that the licensee adequately implemented its primary containment leakage rate testing program (i.e., Types A, B, and C leakage tests) for the MNGP containment. The results of past ILRTs and recent LLRTs demonstrate acceptable performance of the MNGP containment and demonstrate that the structural and leaktight integrity of the containment structure is being adequately maintained. The NRC staff also finds that the structural and leaktight integrity of the MNGP containment will continue to be monitored and maintained if MNGP adopts NEI 94-01, Revision 2-A as the 10 CFR Part 50, Appendix J, Option B, implementation document. Accordingly, the NRC staff determined that there is reasonable assurance that the structural and leaktight integrity for the MNGP containment will continue to be maintained, without undue risk to public health and safety, if the current Type A test intervals are extended to 15 years on a permanent basis.

The NRC staff concluded that it is acceptable for MNGP to (i) revise TS 5.5.11, "Primary Containment Leakage Rate Testing Program," to adopt NEI 94-01, Revision 2-A, as the 10 CFR Part 50, Appendix J, Option B, implementation document; and (ii) extend on a permanent basis the Type A test interval up to 15 years.

3.5 Evaluation of Specific Changes to TS 5.5.11

3.5.1 Revision to TS Requirement 5.5.11a

As discussed above, the licensee has adequately addressed the six conditions identified in Section 4.1 of the NRC SE for NEI 94-01, Revision 2-A. Accordingly, the NRC staff finds that MNGP's adoption of NEI 94 01, Revision 2 A, as the implementation document in TS 5.5.11, "Primary Containment Leakage Rate Testing Program," to replace RG 1.163 is acceptable.

3.5.2 Deletion of TS 5.5.11a "Exception" 1

As authorized by Amendment No. 134, TS 5.5.11a "exception" 1 refers to a one-time change extending the interval after the Type A test performed in March 1993 from at least once per 10 years to at least once per 15 years. The last Type A test was performed in April 2007. Therefore, the subject interval is complete. Accordingly, the NRC staff finds this change acceptable.

3.5.3 Revision to the First TS 5.5.11a "Exception "2

With the deletion of TS 5.5.11a 1, the previous TS 5.5.11a "exception" 2 has been renumbered TS 5.5.11a 1. This is an administrative change and is, therefore, acceptable.

NEI 94-01, Revision 2-A, replaces RG 1.163 as the 10 CFR Part 50, Appendix J, Option B, implementation document. NEI 94-01, Revision 2-A, recommends the test guidance of ANSI/ANS 56.8-2002 instead of the test guidance of ANSI/ANS 56.8-1994, as recommended by NEI 94-01, Revision 0.

The proposed changes to this TS requirement replace the references to ANSI/ANS 56.8-1994 and NEI 94-01, Revision 0, with ANSI/ANS 56.8-2002 and NEI 94-01, Revision 2-A, respectively. However, since the main steam line pathway leakage contribution remains excluded from the sum of the leakage rates from Type B and Type C tests, the intent and scope of this TS requirement remain unchanged. Accordingly, the NRC finds the change to this TS requirement acceptable.

3.5.4 Revision to the Second TS 5.5.11a "Exception" 2

NEI 94-01, Revision 2-A, replaces RG 1.163 as the 10 CFR Part 50, Appendix J, Option B, implementation document. NEI 94-01, Revision 2-A, recommends the test guidance of ANSI/ANS 56.8-2002 instead of the test guidance of ANSI/ANS 56.8-1994, as recommended by NEI 94-01, Revision 0.

The proposed changes to the TS requirements replace the references to ANSI/ANS 56.8-1994 and NEI 94-01, Revision 0, with ANSI/ANS 56.8-2002 and NEI 94.01, Revision 2-A, respectively. However, since the main steam line pathway leakage contribution remains excluded from the sum of the leakage rates from Type A tests, the intent and scope of this TS exception remain unchanged. Accordingly, the NRC finds the change to this TS requirement acceptable.

3.5.5 TS Evaluation Conclusion

The NRC staff reviewed the proposed change to verify the revised program description continues to contain the appropriate administrative controls for the containment leak rate testing program. The staff concludes that the revised TSs continue to provide the appropriate administrative controls to ensure that the requirements of 10 CFR 50.36(c)(5) continue to be satisfied.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified on April 11, 2017, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (81 FR 24663). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 25, 2017

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 5.5.11, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NO. MF7359) DATED APRIL 25, 2017

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